(H'(0.07)), recognizing that this is an overestimation of the true shallow dose. Until further research is conducted, this assumption is considered reasonable.

2.3.1.3 Uncertainty

There has been very little published about the uncertainty of beta dosimetry. Langmead and Adams (1967) compared two types of film badges and reported that for beta exposure the accuracy ranged from –25% to +60%. Considering the similar mechanisms between photon and beta film dosimetry, the methodology discussed in section 2.1.1.3 should be applied.

2.3.1.4 Example

Table 2.4 below is an example of a worker's beta dosimetry results from the Hanford facility in 1963. The estimated uncertainty was calculated using the simplified methodology discussed in section 2.1.1.3.3 with a detection limit of 30 mrem and a standard error of ± 30%.

Table 2.4 Example of beta dosimeter dose and uncertainty

Date	Beta Dose (mrem)	Standard deviation $\sigma(E)$
1/25/1963	39	19
2/22/1963	0	
3/22/1963	59	23
4/19/1963	0	
5/17/1963	84	29
6/14/1963	0	
7/12/1963	86	30
8/09/1963	0	
9/06/1963	89	31
10/04/1963	0	
11/01/1963	0	
12/27/1963	0	

The central dose is 357 mrem with a standard deviation of 60 mrem. Figure 2.10 provides the estimated electron dose distribution.

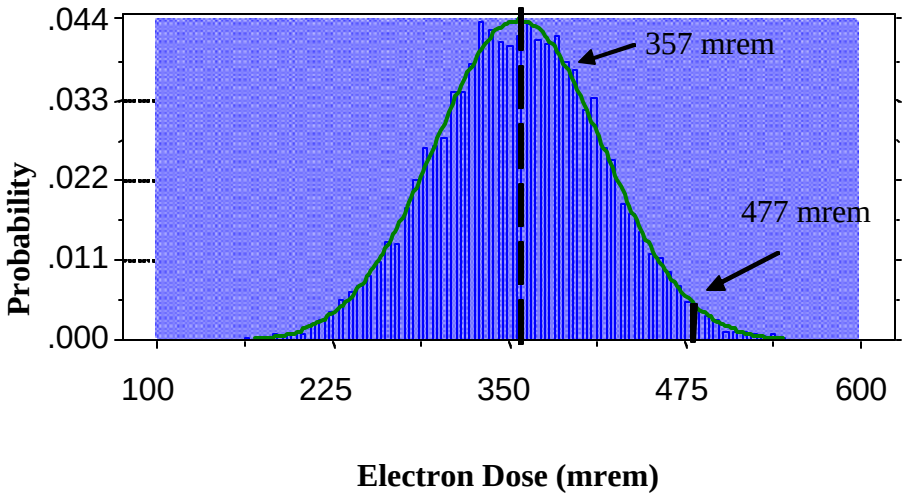

Figure 2.10 Example of electron dose distribution with a mean of 357 mrem and an upper 95% confidence interval of 477 mrem.

2.3.2 Missed Dose

2.3.2.1 Background

As with all dosimeters, electron doses below the limit of detection were not recorded. In addition, many facilities measured the electron or beta dose but did not record this dose in the individual's annual dose of record. The combination of doses below the limit of detection in conjunction with early reporting criteria can result in significant missed dose.

2.3.2.2 Method

The missed dose will be calculated using the same method as that for photon and neutrons. The LOD/2 should be applied for each zero reading. The summation of the LOD/2 doses will produce the central dose estimate.

2.3.2.3 Uncertainty

The uncertainty for missed electron exposures is also assumed to follow a log normal distribution with the upper 95% confidence interval being the LOD times the number of zero readings.

2.3.2.4 Example

Using the data from example 2.3.1.4, the limit of detection was 30 mrem and there were 7 zero monthly measurements. The central tendency of the missed dose distribution would be 105 mrem, with an upper 95% confidence interval of 210 mrem. This distribution is shown in figure 2.11.

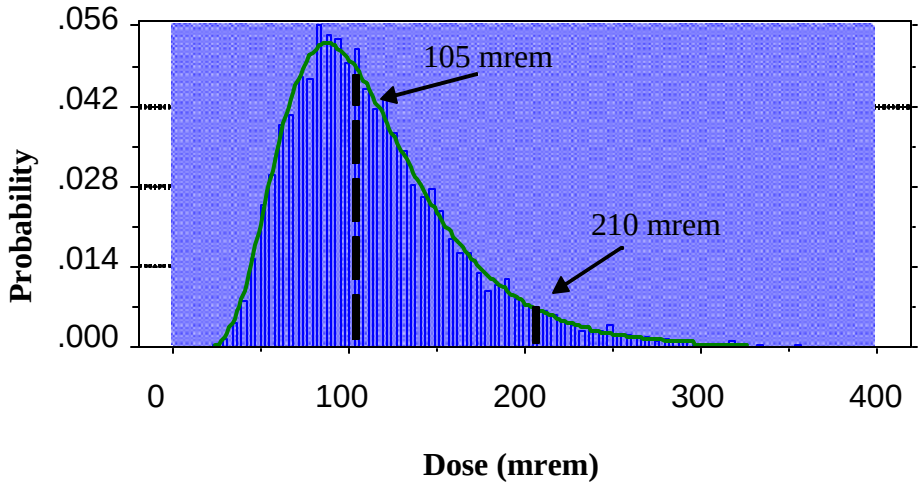

Figure 2.11 Electron missed dose distribution with a geometric mean of 105 mrem and an upper 95% confidence interval of 210 mrem.

2.3.3 Skin Contamination

2.3.3.1 Background

Skin contamination can result in significant electron exposures. While frisking out of contamination areas, some workers might have triggered alarm levels such that decontamination of the skin was necessary. These skin contamination incidents have typically been recorded in the individual's radiological exposure records.

2.3.3.2 Method

2.3.3.2.1 Location of Contamination

To be included in the skin dose, the contamination must have occurred on a body part where the skin cancer originated. For instance, a worker diagnosed with skin cancer on his shoulder has an incident report where contamination was noted on his shoes. The contamination on the shoes should not be calculated into the skin dose on the shoulder. On the other hand, if a worker was found to have contamination of their coveralls over their shoulder, the dose from the skin contamination should be included in the dose estimate. Unfortunately, on some reports, the location of the contamination is not precisely described. In these instances, to be claimant friendly, the contamination should be assumed to be on the cancer site.

2.3.3.2.2 Dose Calculation

For calculating the dose from skin contamination, a program such as VARSKIN[1] can be used to estimate the skin dose. The default skin depth should be 0.07 mm. If the area of the skin cancer is known, the dose should be calculated for that surface area. If the skin

[1] This is not an endorsement of the VARSKIN program, and is presented as one example of a typical program that could assist in skin dose computations.

cancer area is unknown, the contamination area, if known, should be assumed to be the surface area of the skin cancer, however the surface area should not be less than 1 cm^2. The shielding effect of any personal protective equipment such as coveralls, gloves, plastics, etc. worn should be considered if known.

2.3.3.3 Uncertainty

When conducting dose reconstruction for skin cancer, there are multiple parameters which must be taken into consideration such as the activity, average area of the measurement probe, average area of the actual contamination, etc. Professional judgment should be used to determine the most probable exposure parameters in arriving at the central tendency of the log normal distribution of the dose. The maximum or 95% dose limit should be calculated assuming the most reasonable claimant friendly assumptions such as a minimum surface area of 1 cm^2, no protective clothing, negligible distance between contamination and skin, etc.

2.3.3.4 Example

A worker at a reactor facility, after 2 hours in the contamination area, was found to have 400,000 dpm (0.18 μCi) of beta-gamma contamination on the shoulder of his coveralls. Fuel particle is assumed to be the isotope, an air gap thickness of 1 mm, the coverall thickness is assumed to be 0.7 mm and a density of 0.4 g/cm2 (default values of VARSKIN). Assuming the worker picked up the contamination at the midpoint of his work, the dose to the skin would be 89 mrad. The maximum dose would be 598 mrad assuming an air gap of 1 mm, no protective clothing, and an exposure time of 2 hours. Using a log normal distribution, the skin dose distribution is depicted in Figure 2.12.

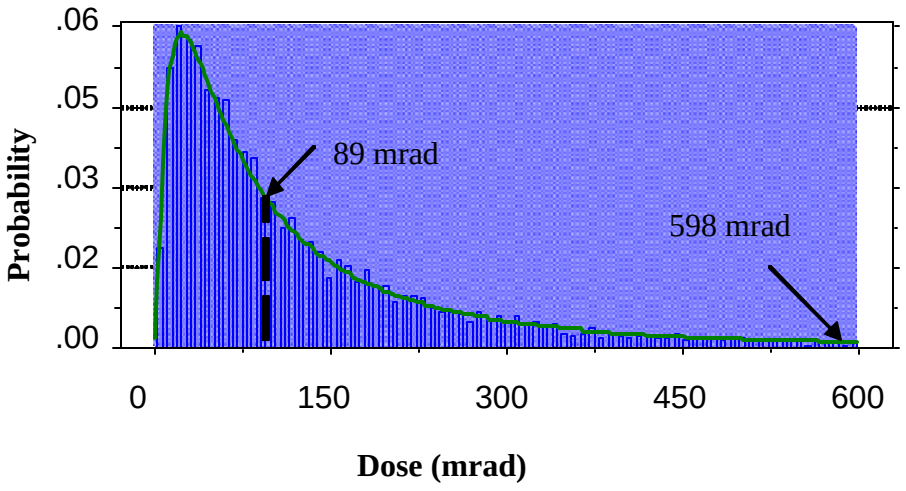

Figure 2.12 Electron dose distribution from skin contamination incident with a geometric mean of 89 mrad and an upper 95% confidence interval of 598 mrad.

3.0 EXTERNAL DOSE RECONSTRUCTION – INCOMPLETE, MISSING OR NO MONITORING DATA

Incomplete or missing personal monitoring usually occurs either between two periods of monitoring data or at the beginning or end of a monitoring time period. When personal monitoring data is missing between two other periods of monitoring, interpolation between the two-monitored time periods may be reasonable. When the incomplete data is either before or after monitoring data, extrapolation may be reasonable, however caution should be used to properly account for any trends that may exist.

3.0.1 Interpolation of Missing Personal Monitoring Data

In some instances, dosimetry records might be missing for a portion of an individual's work history. However if the individual has sufficient monitoring data prior to and after the missing data, the dose can be interpolated by a simple average between the two monitoring periods. The interpolation would be considered reasonable providing the work practices, radiological protection measures, and the administrative and engineering controls did not change. In addition, interpolation may be conducted only if there is no indication, whether from the claimant or site radiological records, that a radiological incident resulting in a higher exposure occurred during the time period of missing data.

3.0.2 Extrapolation from Incomplete Personal Monitoring Data

At some sites, as the radiological monitoring practices were being developed, early dosimetry was rather crude and not all external radiation types were measured. As radiological monitoring programs became more sophisticated, more radiation types and energies were measured and recorded in personal monitoring records. Most programs started with measurements of high-energy photons and then added beta or electron measurements followed by neutrons. In order to reconstruct an individual's dose during these early time periods, some extrapolation from adjacent (near-by) time periods may be necessary. Caution must be used, however, to account for trends in exposure data resulting from differences in work practices, implementation of radiological, administrative, and/or engineering controls that might change the exposure pattern.

Uncertainty from either interpolation or extrapolation could be very difficult to accurately determine. Therefore claimant friendly upper bounds should be used.

3.0.3 No Personal Monitoring Data

When no personal monitoring data is available, the external radiation dose should be reconstructed based on 1) co-worker data, 2) radiation survey data or 3) source term information. As noted in section 1.4, Dose Reconstruction - Hierarchy of Data, co-worker data should be used prior to radiation surveys and survey data should be used before source term information. It should be recognized that dose reconstructions based on survey data will probably be biased, since monitoring practices tended to be recorded at the highest level to ensure compliance, but this is an acceptable bias in a claimant friendly compensation program. If no survey data is available, the dose should be estimated based on the activity of the source term, engineering and administrative controls, and work history.

3.1 Photon Exposures

3.1.1 Photon Dose Reconstruction – Co-worker Data

At some facilities, only a subset of workers was monitored for radiation exposure to demonstrate compliance with orders or regulations. In these instances, the claimant has been asked during the CATI for a list of co-workers who worked with the claimant. Data from the claimant's co-worker(s) should be used when monitoring data is incomplete or missing. In some instances, multiple co-workers were monitored and an average was reported for the remainder of the group. The benefit of the doubt should be given to the claimant and the maximum reasonable worker dose within the group should be used. Since dosimetry data is being used, the methods discussed in Section 2.1 should be used for dose reconstructions.

3.1.2 Photon Dose Reconstruction – Survey Data

3.1.2.1 Background

Throughout the history of radiological operations, radiation surveys using ionization chambers, Geiger-Mueller detectors, and scintillation detectors have been conducted on a routine basis at most weapons production plants. These data have typically been reported in radiation survey reports or on radiation work permits in units of exposure rate or dose rate such as mR/hr, mrem/hr, etc. These data, in conjunction with the duration of exposure, should be utilized only when personal monitoring data is not available, however they should be used before source term data.

3.1.2.2 Method

The exposure or dose can be calculated by simply multiplying the exposure or dose rate by the duration of exposure or dose.

$$Dose = \dot{D} \times t$$

Exposure rate, in units of roentgen per hour (R/hr), has been reported on most early survey data sheets. In later years, when dose rate was reported, some consideration for the method of calibration of the instrument is necessary, although most will be ambient dose equivalent. Also, caution should be used to ensure the reported dose is not a shallow dose (i.e. open window).

From area survey data, the exposure rate was generally well known and access to areas with very high exposure rates was typically restricted. In addition, since most radiological jobs ***do not*** result in exposure 8 hours a day, 5 days a week and 50 weeks per year, time is one of the most important variables. For ease of calculation, time should be divided into hours, days and weeks for a given year. At some facilities, the environmental dose discussed in section 2.4 can be used as a reasonable estimate if no other data is available.

3.1.2.3 Uncertainty

Generally, when using survey data for dose reconstruction there are only two variables, the distribution of measurements in the work area, and the duration of the exposure. Unlike dosimetry measurements, the uncertainty associated with survey data will tend to result in rather large standard deviations. If a normal distribution were used, the lower bound could be less than zero in some cases. Since sampling is conducted for the final distribution, the log normal distribution is believed to be the most reasonable uncertainty distribution based on survey data.

3.1.2.4 Example

A worker estimates exposure 6-8 hours per day 4-5 days a week for a three-month period (13 weeks). The resulting time of exposure would be approximately 410 hours. The survey data during this time period indicates an average exposure rate of 1.5 mR/hr, the worker's average exposure would be 615 mR. If the maximum exposure rate in the area was 2.5 mR/hr, the corresponding upper 95% confidence interval would be 1300 mR.

3.1.3 Photon Dose Reconstruction – Source Term

3.1.3.1 Background

Dose reconstruction from a source term is relatively difficult and the associated uncertainty is relatively large. Before conducting a dose reconstruction based on source term data, an investigation should be conducted to determine if the process is sufficiently similar to another operation at a monitored facility such that other worker data or survey data could be used to estimate workplace exposure levels. When worker and survey data are not available and source term data is used for the dose reconstruction, all assumptions and parameters used in the calculation must be clearly stated and documented in the dose reconstruction report.

3.1.3.2 Method

The source term (S) can sometimes be determined through process or material receipt records, if available. However, facility-handling information is critical to determine the approximate time, distance, and shielding assumptions needed to adequately calculate a dose to a worker. The general point source equation for calculating external exposure based on source term information is:

$$Dose = \frac{SD(E)}{4\pi r^2} e^{-\mu r} \times t$$

S = Source Strength or activity (gamma ray emission)

$D(E)$ = Flux to doserate conversion factor

$e^{-\mu r}$ = shielding component

r = distance between worker and external source

t = duration of exposure

This equation is the simplest form of the attenuation coefficient (e^{-mr}) with no buildup factor. Generally, more complex calculations are required to account for the effects of geometry, self-shielding, multiple shielding and buildup.

Computer programs such as Microshield[2] can greatly facilitate the computations from source term information. In addition, these programs also enable some worst-case examples to be developed to bound the uncertainty of the most reasonable estimate. Received

3.1.3.2.1 Source Term
Generally, the source of the radiation exposure can be identified from material receipt or processing records. Within these records there may be information on the quantity and/or size of the material. With knowledge of density, purity, isotopic content, weight and/or dimensions of the material, the quantity of activity can be calculated from health physics first principles.

3.1.3.2.2 Average Energy
Most radionuclides emit multiple gamma and x-rays at varying yields per disintegration. Since IREP uses three photon energy intervals and five neutron energy intervals, the energy of the emissions can be grouped accordingly and the yields determined by group.

3.1.3.2.3 Time of Exposure
As with the survey data dose reconstruction, the time or duration of the exposure is one of the most critical factors to be estimated. As with dose reconstruction using survey data, specific information on duration of exposure expressed as hours per day, days per week, and weeks per year will assist in a more accurate estimate of exposure duration.

3.1.3.2.4 Distance from Source
The distance is another important parameter in estimating exposure to a radioactive material. At some facilities, workers were separated by tens of feet from radioactive materials due to engineering or administrative controls, while at other facilities, workers handled radioactive materials in bench top experiments such that the distance from the source was approximately 18 inches. Since exposure rate decreases as the square of the distance, this parameter also can have a significant impact on the estimated dose.

3.1.3.2.5 Shielding
For high-level sources, shielding was generally used as an engineering control to protect workers from excessive radiation exposure. In addition, some high-density radioactive materials such as uranium also shield a significant portion of the photons emitted.

3.1.3.3 Uncertainty

Dose uncertainty from source term estimates is relatively large. The most reasonable parameters of source strength, average distance, exposure duration, and shielding should

[2] This is not an endorsement of the Microshield program, and is presented as one example of a program that could assist in the dose computations.

be used to compute the most likely dose. Each of these parameters should then be reasonably estimated to maximize the dose (claimant friendly). Assuming a normal distribution, the most likely estimate should be the mean with the upper 95% limit being the claimant friendly estimate.

3.1.3.4 Example

For example, suppose a worker is measuring the diameter of 5% enriched uranium fuel rods using a caliper. Each cylindrical fuel rod is 6 inches in length and on average 1.5 inches in diameter, with a 1/8 inch aluminum jacket surrounding the rod. The source strength would be relatively constant; however, the distance from the rod would vary between 6 to 18 inches with the most likely being the midpoint of 12 inches. The claimant indicates he conducted this work on average for 6 hours a day 3-4 days a week for a six-month period. The most likely dose estimate is calculated using the midpoints. The upper 95% confidence interval of the estimate should be estimated based on 8 hours a day for 4 days a week during the six-month period at a distance of 6 inches.

3.1.4 Photon Dose Reconstruction – Control Limits

3.1.4.1 Background

Dose reconstruction based only on administrative or radiological monitoring controls will result in a gross overestimation of the claimant's dose. Unfortunately, if no monitoring records of any type can be found and the source term is unknown, an upper external dose estimate can be developed using occupational radiation protection limits. This of course assumes that appropriate controls were in place to prevent exposures in excess of occupational limits. When conducting a dose reconstruction using control limits, all assumptions must be clearly stated in the dose reconstruction report.

3.1.4.2 Method

There are three radiological control limits that can be used for dose reconstruction: threshold for required monitoring; radiation posting limits; and annual radiation dose limits.

3.1.4.2.1 Monitoring Not Required

This method is most appropriate for office workers who were not monitored due to the low potential for exposure. In these instances, the central point estimate should be the threshold level for monitoring. At most facilities, this value was 100 mrem/year.

3.1.4.2.2 Posted Control Limits

This method is most appropriate for short duration exposures when an unmonitored person entered a radiological controlled area without proper monitoring. In these instances, the midpoint dose rate between posted areas should be used as a reasonable estimate. This midpoint dose rate multiplied by the number of hours of exposure will provide the central dose estimate. The upper bound of the posted area multiplied by the number of hours in the areas will result in the upper 95% dose estimate.

3.1.4.2.3 Annual Radiation Dose Limits

Dose reconstruction using annual limits is relatively simple. The dose assigned is the maximum allowable monthly dose times the number of months worked. Since a worker could have received the annual limit in a short time frame, the acute exposure should be used. Since dose reconstruction using annual limits will yield unreasonably large exposure estimates some restrictions apply to the use of annual limits. The method should only be used for short employment durations of less than one year and for a maximum dose of 5000 mrem.

3.1.4.3 Uncertainty

As indicated in section 3.1.3.4, the midpoint of the dose range should be used as the most likely estimate with the maximum being the upper 95% of a lognormal distribution. Although DOE orders have specified, weekly, monthly, and quarterly dose limits, workers have been allowed to exceed these administrative limits as long as they did not exceed the annual limits. Generally the central estimate a dose distribution can be developed using the weekly, monthly, or quarterly exposure limit with the upper 95% confidence interval being the annual radiation dose limit. However, when the annual radiation dose limit is used for dose reconstruction, this dose should be considered the maximum dose. Therefore a constant should be used and thus there is no distribution.

3.1.4.4 Example

The examples provided below describe using posted control limits and annual radiation dose limits to estimate a workers radiation dose.

3.1.4.4.1 Posted Control Limit Example

A worker enters a radiation area without wearing a dosimeter, and radiation survey data for this time period is not available. The radiological protection requirements for the work era indicate that the minimum dose rate for a posted radiation area was 5 mrem/hr with a maximum of 100 mrem/hr. The worker was in the area for approximately 4 hours. The most likely dose would be 20 mrem and the upper 95% dose would be 400 mrem.

3.1.4.4.2 Annual Radiation Dose Limit Example

A claimant indicates they worked with radioactive materials for approximately 2 months at an unmonitored facility in 1974 and no source term information is available. The maximum allowable dose (Radiation Dose Limit) was 5000 mrem/year, thus the maximum monthly dose rate would be 417 mrem/month. The most likely dose would be 833 mrem, with an upper 95% limit of 5000 mrem.

3.2 Neutron Exposures

As with photon exposures, estimating neutron exposures without personal monitoring data is relatively difficult. The three main types of data to be used are: 1) co-worker data, 2) survey data, or 3) source term data. Generally neutron exposures are accompanied by photon radiation. As a result, radiation control limits have combined these doses for administrative control of radiological areas. Therefore, neutron exposures should never be estimated based on radiation dose limits.

3.2.1 Neutron Dose Reconstruction - Co-worker Data

After individual monitoring data, co-worker data is considered the next most accurate indicator of exposure. This data should be used whenever individual monitoring data for the claimant is not available. When group (co-worker) data is available, the benefit of the doubt should be given to the claimant and the maximum worker dose within the group should be used. Since dosimetry data is being used, the methods discussed in Section 2.2 should be used for dose reconstruction from co-worker data.

3.2.2 Neutron Dose Reconstruction – Survey Data

3.2.2.1 Background

Throughout operations at nuclear weapons sites, neutron monitoring has been conducted using proportional counters such as BF_3 detectors and recently, tissue equivalent proportional counters (TEPC). Around nuclear reactors, neutron measurements have been conducted to verify adequate shielding of the reactor, thus survey data should be available to estimate exposures. At one facility, neutron-monitoring data has been found for glovebox lines in chemical separations areas back into the late 1940's (Reddie and Whipple, 1949). This data in conjunction with average stay times can be used to estimate exposures.

3.2.2.2 Method

The general equation is the same as described in section 3.1.2 and is provided as follows.

$$Dose = \dot{D} \times t$$

where: $\dot{D}$ = dose rate or fluence
t = duration of the exposure

Generally, an average of the dose rate measurements in the workplace should be used for the central estimate, however, some consideration should be given for the most reasonable measurements. For example if dose rate measurements are taken throughout a room where a claimant worked, but the highest measurements were recorded near the gloveboxes. The worker indicated he spent most of his time in the room near the gloveboxes, the measurements closest to the gloveboxes should be used instead of the average dose rate measurements in the room.

Depending on the instrumentation used, either the dose rate or the fluence is typically reported. The fluence allows for easy conversion to organ dose and should be used whenever possible. When dose rate is reported, some additional information on the quality factor is needed to convert to an absorbed dose before the conversion to organ dose can be conducted.

3.2.2.3 Uncertainty

As with most exposure discussions in this guide, the central estimate should be an average of the survey data, with consideration for the most reasonable estimate. The

upper bound should be estimated by applying the maximum work time period with the maximum recorded dose rate for the area. Generally, survey data follow a lognormal distribution, therefore this distribution should be used for the uncertainty distribution. In addition, since the uncertainty is expected to be relatively large, a significant percent of the data could be negative if a normal distribution were used. Therefore the normal distribution is not recommended.

3.2.2.4 Example

A chemist works with a plutonium solution in a glovebox for a quarter (13 weeks). At 18 inches from the surface of the glovebox, the fast neutron flux was measured to be 12 n/cm^{2}s. At the surface of the glovebox, the flux was measured to be 35 neutrons/cm^{2}s. On average, a worker stood approximately 18 inches from the face of the glovebox, for 4-6 hours per day for 2-4 days a week. The central exposure estimate would be 8.42 x 10^6 neutrons/cm^2 and the upper 95% would be 3.93 x 10^7 neutrons/cm^2. Assuming an average neutron energy of 4 MeV, the ambient dose equivalent would be approximately 344 mrem. The upper 95% estimate would be 1604 mrem (Figure 3.2).

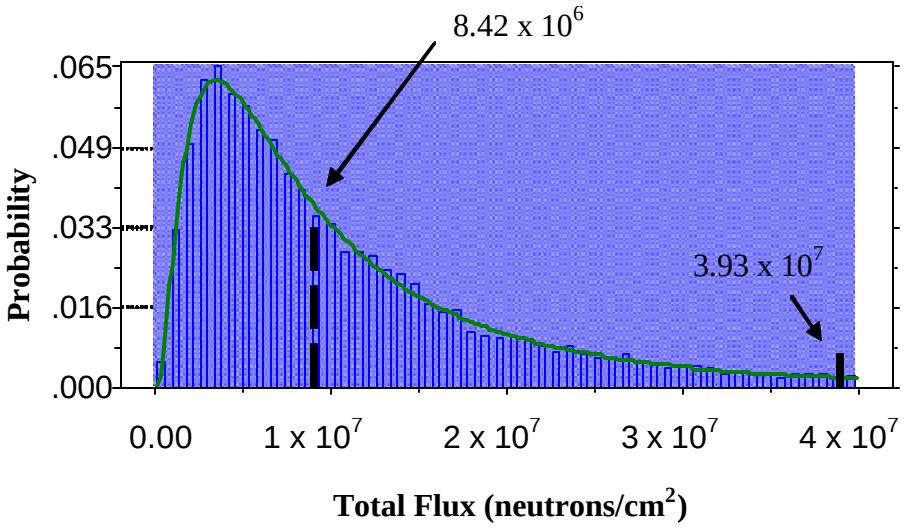

Figure 3.2 Estimated neutron exposure distribution for chemist. Geometric mean of 8.42 x 10^6 n/cm^2 and an upper 95% confidence interval of 3.93 x 10^7 n/cm^2.

3.2.3 Neutron Dose Reconstruction - Source Term Data

3.2.3.1 Background

Dose reconstruction from a neutron source term should only be conducted when no survey data is available and relatively simple exposure geometries are appropriate. NCRP 38 (1971) provides general guidance for radiological protection against neutron radiation.

3.2.3.2 Method

This methodology described in NCRP 38 (1971) should be used when estimating neutron exposures from various shielded sources. The general point source equation is similar to the photon point source equation in section 3.1.3, however the attenuation coefficient is replaced with a neutron removal cross section. The basic principle of the removal cross section is that the neutron is scattered and then either absorbed by the material along the path between the source and the dose point or undergoes additional scattering away from the dose point. NCRP 38 discusses criteria for which the removal cross section is a valid assumption.

$$Fluence = \frac{S}{4\pi r^2} e^{-\Sigma_R x} \times t$$

S = Source Strength (neutrons/sec)
$e^{-\Sigma_R x}$ = shielding component
r = distance between worker and source
t = duration of exposure

Additional removal cross sections can be calculated using the methodology discussed in NCRP (1971) Report 38.

3.2.3.3 Uncertainty

The uncertainty associated with dose estimation from source term data is relatively large and could vary by an order of magnitude or more. As with the photon measurements there are several sources of uncertainty; including the duration of the exposure, the distance from the source, variations in the shielding thickness, and the uncertainty of the initial neutron fluence. The most reasonable value of each parameter should be used to determine the central estimate, while claimant friendly assumptions should be made to estimate the upper bound of the distribution. Generally, a normal distribution should be applied, however if the upper bound uncertainty (2σ) subtracted from the central estimate is less than zero, a lognormal distribution should be used.

3.3 Electron Exposures

Electron exposures are only important for certain cancer sites such as the skin, breast, and possibly for the testes, depending on the electron energy and shielding. Electron exposures do not need to be calculated for deep organs.

The use of co-worker data can be used providing there were no contamination incidents and only non-extremity dosimetry is used. In the absence of co-worker data, survey data can be used, however, a thorough understanding of the measurement data is needed to adequately interpret the dose since much of the data is reported in units of activity and not external dose rate. Source term data can also be used, however, great care should be given to the distance from the source and the duration of the exposure since beta dose rates are greatly diminished a few centimeters from the source. Generally, administrative

dose limits for skin exposures are very large, however contamination control limits could be used to estimate the upper bound of low-level exposures for initial dose assessment.

3.3.1 Electron Dose Reconstruction - Co-worker Data

Unlike photon and neutron radiation, electrons have very low penetrating ability. Due to these physical properties, co-worker data is of limited value for electron exposures. Generally only standard dosimetry would be a good measure of exposure. Differences in job functions, proximity to the source and duration of exposure make extremity dosimetry highly uncertain and should not be used, unless the identical job function is performed, and the proximity to the source is identical and relative fractions of exposure time can be clearly established. Co-worker skin contamination incidents should not be applied for dose reconstruction.

3.3.2 Electron Dose Reconstruction - Survey Data

3.3.2.1 Background

Open window GM detectors or thin window ionization chambers have been used to measure the beta dose rate, however, in some instances, only contamination survey data is available in units of activity. The method section is subdivided into dose rate surveys and contamination surveys.

3.3.2.2 Method

3.3.2.2.1 Electron Dose Rate Data

Electron or beta dose rate survey data in conjunction with duration of exposure can be used to estimate electron dose, using the standard equation discussed in section 3.1.2 and 3.2.2.

$$Dose = \dot{D} \times t$$

where: $\dot{D}$ = dose rate usually in mrad/hr
t = duration of the exposure

3.3.2.2.2 Contamination Survey Data

In some instances contamination survey data could be used to estimate the beta dose rate. For these computations, the computer program VARSKIN may be used as it integrates the Berger (1971) point kernel equation. The computational methods and details can be found in NUREG/CR-5873 (Durham, 1992). Basic inputs to VARSKIN include source geometry, activity, source size, air gap, protective layer thickness, and density of the protective layer. While the VARSKIN program was designed for skin contamination, by varying the air gap, it can be utilized for external electron skin doses. When utilizing contamination survey data, a large disc source is recommended and minimum averaged dose area should be no less than 1 cm^2.

3.3.2.3 Uncertainty

As with previous uncertainty calculations, the average reading or most likely reading for dose rate measurements or activity measurements should be used as the central estimate. The highest recorded value should be used to calculate the upper 95% bound. The duration should also be varied to determine the upper 95% bound of the log normal distribution.

3.3.2.4 Example

A claimant with skin cancer originating on their chest indicates they once worked for about 2 hours with a section of ductwork that was heavily contaminated with uranium. The CATI indicates they wore coveralls during this work and that survey data was collected. Upon investigation, survey data indicated an average activity of 500,000 dpm/100cm2 with a maximum activity of 2 x 10^6 dpm/100 cm^2. Using default coverall values in VARSKIN, and assuming the average distance between the source and the skin is 30 cm (≈1 foot), the central dose estimate would be 7 mrad for the 2 hour exposure with a maximum dose 66.6 mrad assuming a distance of 10 cm (Figure 3.3).

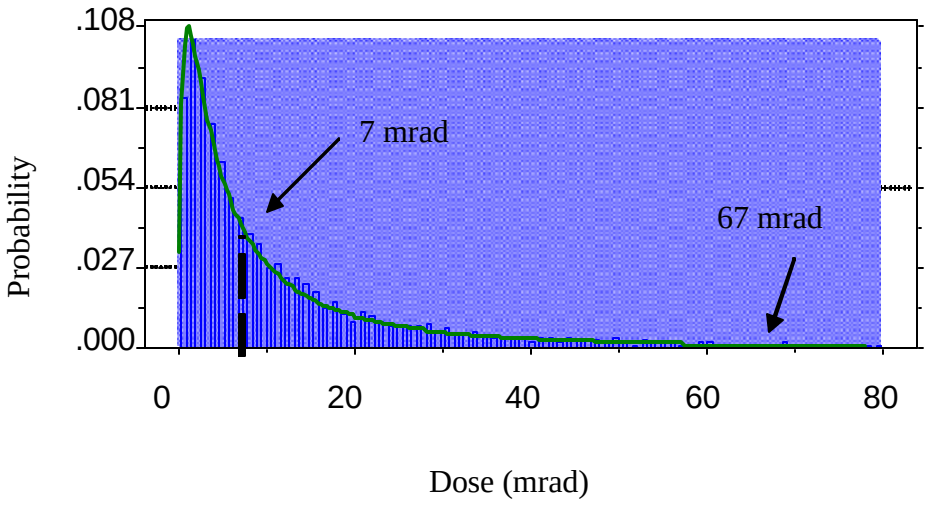

Figure 3.3 Example of skin dose distribution from uranium work with a geometric mean of 7 mrad and an upper 95% confidence interval of 67 mrad.

Clearly this is a low level exposure, however, this method demonstrates that reasonable estimates can be developed from limited exposure information.

3.3.3 Electron Dose Reconstruction - Source Term

3.3.3.1 Background

Electron exposures from source term data are extremely difficult to calculate. This type of dose reconstruction should not be conducted unless detailed information about the source, encapsulation, duration of exposure or contamination levels are known or can be

adequately bounded. The most applicable scenario would be to use this method for unencapsulated bare metal such as uranium.

3.3.3.2 Method

As with the surface contamination methodology, the skin dose rate can be calculated from various geometries using source term activity and a program such as VARSKIN. The dose rate can be combined with exposure time to calculate the central dose estimate as shown in the following equation.

$$Dose = \dot{D} \times t$$

where: $\dot{D}$ = dose rate usually in mrad/hr
t = duration of the exposure

For multiple skin contamination incidents, the sum of the individual incidents in a year will comprise the total skin dose for that year.

3.3.3.3 Uncertainty

As with other source term dose reconstructions, the time, distance, and shielding can be varied to develop the upper dose limit. The electron dose distribution is assumed to follow a log normal distribution. Professional judgment should be used to estimate the most probable exposure, with claimant friendly and clearly stated assumptions, such as no shielding, close distance and maximum exposure time to estimate the 95% upper dose limit. For multiple skin contamination incidents in a single year, the uncertainty should be combined using the square root of the sum of the squares methodology as described in section 2.1.1.3.4.

3.3.4 Example

A claimant with skin cancer on the palm of their hand loaded depleted uranium slugs measuring 6 inches long and approximately one inch in diameter into shipping boxes. The claimant conducted this work intermittently 1-3 hours a day for 3-5 weeks. Through the CATI the claimant indicated he did not wear gloves when handling the uranium. Assuming Pa-234m and Th-234 are in equilibrium with the depleted uranium, the average dose rate is 162 mrad/hr and the maximum contact skin dose rate is approximately 209 mrad/hr. The central tendency parameters would yield a skin dose of 6480 mrad, with an upper 95% dose limit of 15675 mrad (Figure 3.4).

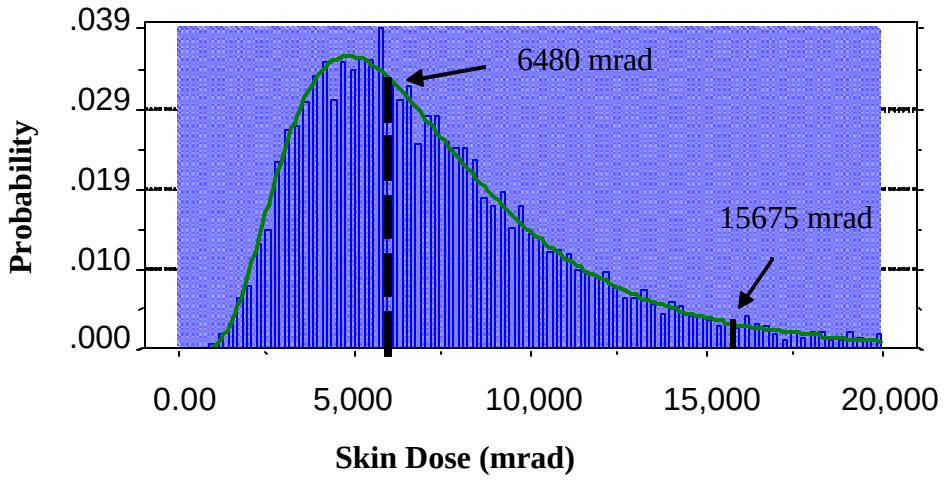

Figure 3.4 Skin dose distribution for worker handling depleted uranium slugs with a geometric mean of 6480 mrad and an upper 95% confidence interval of 15675 mrad.

3.3.4 Radiological Control Limits

Radiological control limits have been used at many DOE facilities to control or prevent the spread of radiological contamination to non-radiological contaminated areas. These limits have been enforced through the use of contamination control checkpoints. Currently there are three levels of radiological contamination postings; radiological buffer area (usually a non-contamination area), contamination area, and high contamination area, which is usually 100 times the non-contamination area upper limit (10 CFR 835). The use of these limits for dose reconstruction is restricted to estimate the upper low-level dose of non-routine radiological workers who might have entered a radiological area for a short time period. The dose assigned from control limits should be limited to a maximum of 5000 mrem.

4.0 CONVERSION TO ORGAN DOSE

The purpose of this section is to provide guidance on the conversion from individual monitoring data to organ dose. For photon exposures, the organ dose conversion coefficients in ICRP 74 convert from free-air KERMA to absorbed dose in the organ of interest. A conversion from monitored dose to free-air KERMA is needed to complete the organ dose conversion. Neutron organ dose conversion factors in ICRP 74 are tabulated per neutron fluence. While some monitoring data has been reported in terms of fluence, traditionally, neutron doses have been reported as either ambient dose at 10mm ($H^*(10)$) or personal dose at 10 mm ($H_{p,slab}(10)$). Since skin is the primary target tissue for electron doses, the dose conversion factors should be calculated for a skin depth of 0.07 mm.

4.1 Photon Dose

The two basic types of data involved in converting photon doses to organ dose are monitored individual doses, and survey or source term dose rate data.

4.1.1 Monitored Exposure/Dose to Organ dose

4.1.1.1 Exposure (R) to free-air KERMA (K_a)

Most early monitoring data was reported in units of exposure and not a deep dose at 10 mm. Using figure 4.3 in ICRU 43 (1988), and the ambient deep dose (H*(10)) from ICRP 74 (1996), the conversion factor from exposure to free-air KERMA can be calculated. Table 4.1 provides the conversion factors used in calculations to develop the organ dose conversion factors.

$$E \rightarrow K_a = \frac{H^*(10)}{K_a} \times \frac{1}{\frac{H^*(10)}{E}} = \frac{E}{K_a}$$

where

$E = Exposure$
$H^*(10) = Ambient\ Dose$
$K_a = free-air\ KERMA$

Table 4.1 Conversion factors used in organ dose calculations.

Photon Energy (MeV)	Ambient Dose Equivalent H*(10) - cSv / Exposure (R)[1]	Ambient Dose Equivalent H*(10) - cSv / free-air KERMA (K_a)[2]	Exposure (R) / free-air KERMA (K_a)
0.015	0.25	0.26	1.04
0.020	0.60	0.61	1.02
0.030	1.00	1.10	1.10
0.040	1.30	1.47	1.13
0.050	1.46	1.67	1.14
0.060	1.55	1.74	1.12
0.070	1.53	1.73	1.13
0.080	1.51	1.72	1.14
0.100	1.43	1.65	1.15
0.150	1.30	1.49	1.15
0.200	1.20	1.40	1.17
0.300	1.13	1.31	1.16
0.400	1.09	1.26	1.16
0.500	1.05	1.23	1.17
0.600	1.04	1.21	1.16
0.800	1.01	1.19	1.18
1.000	1.00	1.17	1.17
2.000	0.96	1.14	1.19
4.000	0.95	1.12	1.18
6.000	0.95	1.11	1.17
8.000	0.95	1.11	1.17
10.000	0.95	1.10	1.16

[1] *Data extracted from Figure 4.3 ICRU 43 (1988)*
[2] *Data from ICRP 74 (1996)*

4.1.1.2 Photon Dose Equivalent H(10) and $H_p(10)$ to free-air KERMA*

Table A.21 in ICRP 74 (1996) lists the conversion coefficients from ambient dose equivalent (H*(10)) to free-air KERMA (K_a) by photon energy. Table A.24 in ICRP 74 (1996) lists the conversion coefficients from deep dose equivalent ($H_p(10)$) to air KERMA (K_a). Once the dose is converted to free-air KERMA, the organ dose is a straightforward multiplication of the dose conversion factors (D_T/K_a) listed in Tables A.2 – A.20 of ICRP 74 (1996).

$$DCF_{Hp(10)\rightarrow D_T} = \frac{1}{\frac{H_p(10)}{K_a}} \times \frac{D_T}{K_a}$$

where:

D_T = Absorbed Dose in Target Tissue

$H_p(10)$ = Personal Dose Equivalent

K_a = free - air KERMA

4.1.2 Area survey or source term data to Organ Dose

Generally, radiation survey data have been reported in units of exposure (R) in free air. For these data, the exposure methodology discussed in section 4.1.1.1 should be used. Area survey dose rates or those calculated from source term information should generally be assumed to be the ambient dose at 10 mm or $H^*(10)$.

4.1.3 Dose Conversion Factor Simplification

The Dose Conversion Factors (DCF) in ICRP 74 are listed by tissue of interest, exposure geometry, and radiation energy. As noted previously, NIOSH-IREP uses energy intervals for the probability of causation calculation. Since ICRP 74 lists the dose conversion factor for multiple energies, some simplification is needed for dose reconstruction under EEOICPA. As shown in Figure 4.1, the dose conversion coefficient is a continuous function of energy. For simplification, the area under the curve from the beginning to the end of the energy interval divided by the range will be used as the simplified dose conversion coefficient. A simple function *(f(E))* was fitted for each energy interval to integrate the area under the curve. The example below is for red bone marrow dose from photons between 30 and 250 keV.

$$DCF_{g,30-200keV} = \frac{\int_{30}^{200} f(E)dx}{Range} = 0.4196 \frac{\text{Bone Marrow - Gy}}{\text{H}_p(10)\ \text{Gy}}$$

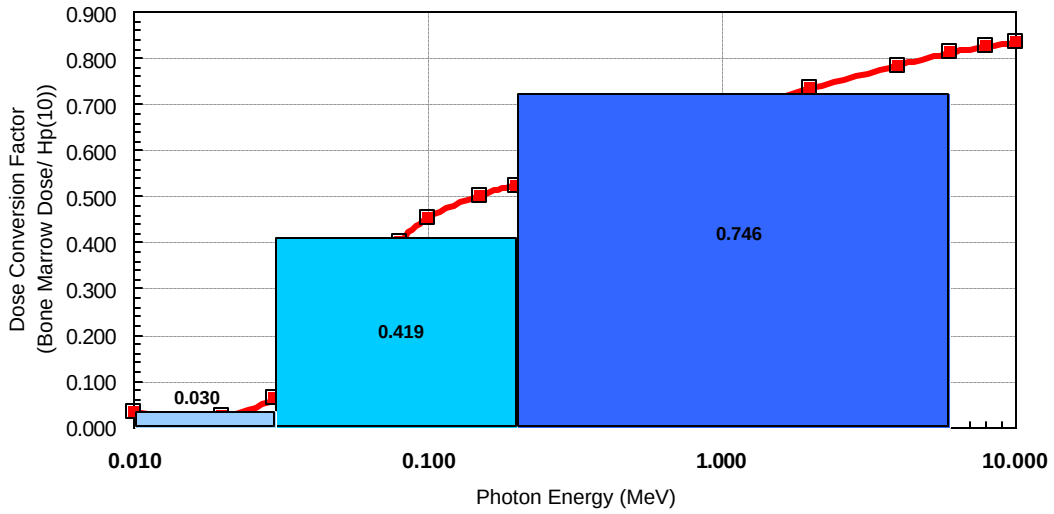

Figure 4.1 Chart of red bone marrow Dose Conversion Factor (DCF) versus photon energy, fitted curve, and associated simplified dose conversion factor for energy band.

Appendix B lists the simplified dose conversion factors by reporting unit (exposure, ambient dose ($H^*(10)$), or deep dose equivalent ($H_p(10)$)) for the three photon energy bands. It should be noted that the upper bound used in the calculation of the high-energy

group for photons is truncated at 6 MeV. This method was employed since there are very few operations at DOE which result in photon exposures greater than 6 MeV.

4.2 Neutron Dose Conversion

4.2.1 Area Monitoring Data to Organ Dose

Area monitoring data has been reported in several different formats. Some earlier measurements report the fluence, with energy information provided, while other measurements are reported in absorbed dose (rad), or dose equivalent (rem).

4.2.1.1 Fluence Data to Organ Dose

When fluence data are provided, the conversion to organ dose is straightforward using tables A.26 through A.40 in ICRP 74 (1996). As with the photon dose conversion factors, the ICRP 74 (1996) tables have been compressed into the five neutron energy intervals for use in the IREP program. These compressed tables can be found in Appendix B of this guide.

4.2.1.2 Ambient Dose (H(10)) to Organ Dose Equivalent*

When ambient dose ($H_S^*(10)$) has been reported (typically in survey data), the site specific quality factor (Q_s) must be removed such that absorbed dose is the fundamental unit. Current ICRP 60 (1990) radiation weighting factors *(w_R)* should then be multiplied by the absorbed dose to develop the standard ambient dose equivalent (H*(10)). From the standard ambient dose equivalent the conversion factors in Appendix B are then applied to determine organ dose equivalent. The conversion from site specific ambient dose to organ dose is illustrated in the following equation.

$$H_T = \frac{H_S^*(10)}{Q_S} \times w_R \times DCF_{H_T / H^*(10)}$$

4.2.2 Personal Monitoring Data to Organ Dose

When routine personal monitoring began, the reported quantity has usually been in dose equivalent. Currently, the standard for personal monitoring neutron data is the deep dose equivalent at 10 mm calibrated using the ICRU slab phantom ($H_{p,slab}(10)$).

4.2.2.1 Neutron Dose Equivalent ($H_{p,slab}(10)$) to Organ Dose Equivalent

Appendix B lists the personal dose equivalent to organ dose equivalent conversion factors. As with the ambient dose, the site specific quality factor (Q_S) should be removed prior to dose calculations and the ICRP 60 (1990) weighting factor *(w_R)* applied before the conversion to organ dose.

4.3 Electron Dose Conversion Factors

ICRP 74 (1996) list energy specific organ dose conversion factor from fluence. It is anticipated that relatively few dose measurements will have been reported in this manner.

ICRP 74 indicates that the dose conversion factor is highly dependant on the electron energy. Since most electron exposures will be a continuum of energies, the site-specific dose conversion factor should generally be used. The shallow dose at 0.07 mm can be assumed to be the skin organ dose.

4.4 Exposure Energy and Geometry

There are six basic exposure geometries discussed in ICRP 74 (1996); the anterior to posterior (AP), posterior to anterior (PA), left lateral (LLAT), right lateral (RLAT), rotational (ROT) and isotropic (ISO) (Figure 4.2). Of these, only four (AP, PA, ROT, and ISO) are of primary interest in dose reconstruction. The AP geometry is the most common geometry experienced by workers who handled radioactive materials. However there are specific job functions in certain types of facilities, which would tend to lead to a different geometry.

4.4.1 Dosimeter and Missed Dose Geometry

For dose reconstruction, professional judgment should be used to determine the most credible geometry or geometry weighting factors *(w_g)* from multiple geometries based upon an individual's work history and the CATI. The work-related Dose Conversion Factor (DCF_W) should be calculated as follows:

$$DCF_W = w_{AP}DCF_{AP} + w_{PA}DCF_{PA} + w_{ROT}DCF_{ROT} + w_{ISO}DCF_{ISO}$$

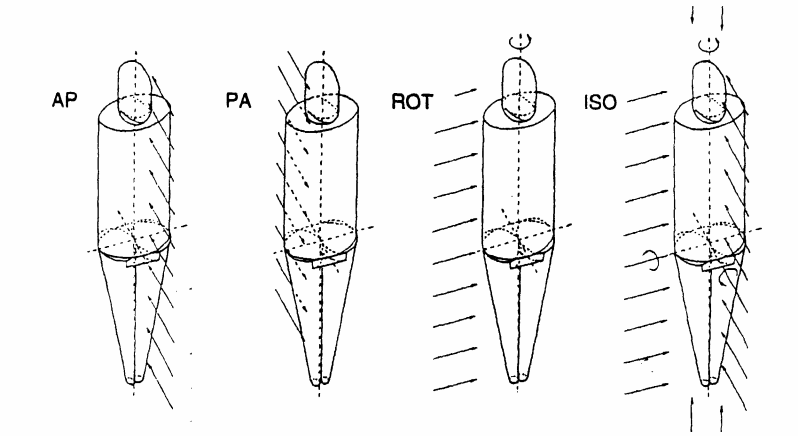

Figure 4.2 Exposure geometries of an anthropomorphic phantom extracted from ICRP 74 (1996)

For example, the isotropic geometry would be reasonable for a general laborer in a uranium manufacturing storage facility, while a lathe worker in the same facility would be more likely to receive the majority of their exposure in an anterior-posterior fashion. A reactor worker refueling a graphite reactor would likely receive their exposure in both the AP and ROT geometry. Table 4.2 provides some general guidance on percentages of exposure geometries.

Table 4.2 Common exposure geometries for various jobs and facilities.

Facility	Job	Geometry	Percentage
Uranium Facility	General Laborer	ISO	75%
		AP	25%
	Machinist	AP	75%
		ISO	25%
	Supervisor	AP	50%
		ISO	50%
Reactor	Fuel Handler	AP	50%
		ROT	50%
	Reactor Operator	ROT	75%
		ISO	25%
Chemical Separations	Glovebox Chemist	AP	90%
		ROT	10%
	Maintenance Worker	AP	50%
		ROT	50%
	Security Guard	ROT	50%
		ISO	50%

4.4.2 Occupational Medical Exposure Geometry

Generally, the exposure geometry for occupational medical (x-ray) exposures is the PA geometry. There are, however, circumstances in which the exposure geometry will be different and these should be applied as appropriate.

4.4.3 Environmental Exposure Geometry

The exposure geometry for environmental doses is almost always isotropic in nature. This assumption should be applied to all environmental doses unless another geometry is more appropriate and has been clearly justified.

4.5 Dose Conversion Uncertainty

4.5.1 Energy Uncertainty

The uncertainty resulting from the energy simplification is assumed to follow a uniform distribution using the dose conversion factor lower and upper bounds within the energy interval for the specific exposure geometry. Table 4.3 provides an example using the bone marrow example with the anterior-posterior geometry for photons (Figure 4.1).

Table 4.3: Photon Bone Marrow Energy Uncertainty using AP geometry

Photon Energy Band	Mean Dose Conversion Factor	Minimum Dose Conversion Factor	Maximum Dose Conversion Factor
< 30 keV	0.030	0.016	0.063
30 – 250 keV	0.479	0.063	0.540
> 250 keV	0.746	0.540	0.834

4.5.2 Geometry Uncertainty

There is often considerable uncertainty as to the position from which the claimant received radiation exposure. As noted in section 4.3, there maybe some information about job function and position of exposure when handling radioactive materials. Since the "true" exposure geometry is almost never known, an uncertainty distribution about

the dose conversion factor is appropriate. Since likely exposure geometry can be calculated for most jobs, a uniform distribution appears to be inappropriate. However, a triangular distribution with the mode being the most likely geometry, the lower bound being the geometry that would result in the lowest organ dose (or dose conversion factor) and the upper bound being the geometry resulting in the highest organ dose (highest dose conversion factor) maybe appropriate. For the bone marrow example previously discussed, and photon exposures in the 30-250 keV range, using the 100% AP geometry, the geometry and energy resulting in the lowest dose conversion factor is the AP geometry at 30 keV (0.063) and the DCF resulting in the highest dose would be the PA geometry at 250 keV (0.791). The resulting distribution is depicted in figure 4.3.

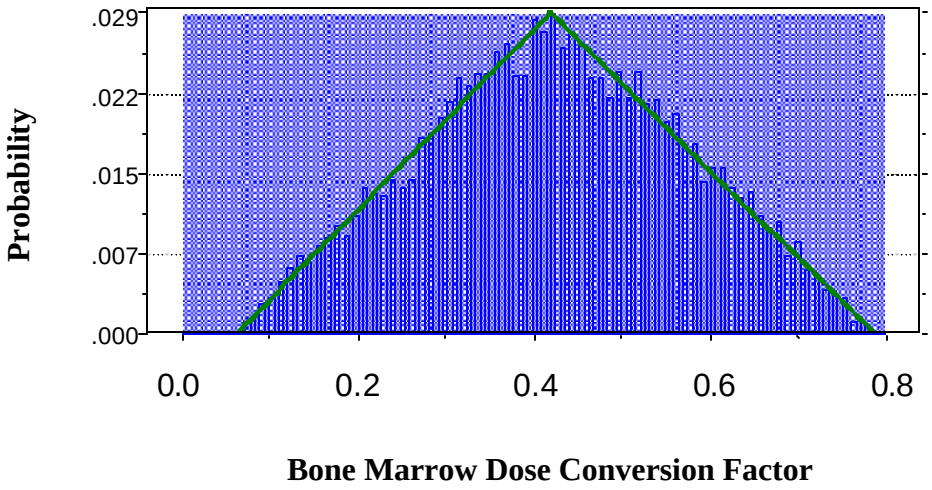

Figure 4.3 Example of the dose conversion factor from $H_p(10)$ dose to red bone marrow dose and intermediate energy photons.

5.0 ANNUAL ORGAN DOSE & DISTRIBUTION

As noted in section 1.5, the final organ dose estimate is compiled for each radiation type and energy. For external exposure it is possible to have a total of 18 different radiation type and energy combinations contributing to the organ doses in a given year. Typically most workers will have fewer than 18, however most will have one to three photon doses depending on the energy. Some might have neutron doses and possibly an electron dose. This section discusses the computation of the dose, the development of the uncertainty distribution, and the final reporting of the dose in an EXCEL file for IREP.

5.1 Organ Dose Computation

5.1.1 Organ Dose Estimate

5.1.1.1 Background

The main purpose of this section is to provide guidance on converting the measured dose into an organ dose and to combine each dose component into a single annual dose estimate for entry into the IREP program.

5.1.1.2 Method

The organ dose for each radiation type and energy are compiled by summing the organ dose components calculated by multiplying the dose or exposure component by the appropriate dose conversion factor. When multiple variations have been reported such as ambient dose equivalent and deep dose equivalent, the conversion should be conducted before the summation. The general equation is as follows:

$$D_{radiation,tissue} = D_D(DCF_W) + D_M(DCF_W) + D_{OM}(DCF_{AP}) + D_E(DCF_{ISO})$$

5.1.1.3 Example

A glovebox chemist who worked in the 200 area at the Hanford facility is diagnosed with leukemia. The high-energy photon dose is calculated by summing the dosimeter dose, the missed dose, and the environmental dose. The worker's occupational medical dose would not be included in the high-energy photon dose but should be included in the intermediate energy photon dose. The worker's dosimeter dose for 1947 was 415 mR with a 95% upper uncertainty of 513 mR (example 2.1.1.4). The claimant was monitored with a film badge for 39 weeks and had 12 positive readings resulting in 27 undetectable measurements (< 30 mrem). The missed dose would be 405 mR with an upper 95% uncertainty of 810 mR. The claimant's environmental dose was 129 mR with an upper 95% uncertainty of 500 mR (example 2.1.4.4). The bone marrow dose is compiled by sampling from each distribution represented as variables in following equation.

$$D_{g,RBM} = D_D(DCF_W) + D_M(DCF_W) + D_E(DCF_{ISO})$$

The claimant's exposure geometry for the dosimeter and missed dose is estimated to be 90% anterior-posterior (AP) and 10% rotational (ROT). This corresponds to a dose conversion factor of 0.721 with a lower bound of 0.570 and an upper bound of 1.007.

The environmental dose is estimated to be 100% from the isotropic geometry, thus the dose conversion factor would be 0.665 with a lower bound of 0.570 and an upper bound of 0.768.

$$\begin{aligned} DCF_W &= 0.90(DCF_{AP}) + 0.10(DCF_{ROT}) \\ &= 0.90(0.720) + 0.10(0.732) \\ &= 0.721 \end{aligned}$$

5.1.2 Uncertainty Distribution

5.1.2.1 Background

The uncertainty associated with the organ dose is computed through random sampling (Monte Carlo) of each distribution used to compute the central organ dose estimate. By using these distributions, the overall organ dose uncertainty can be determined with reasonable precision. It is recommended that 5000 iterations be used to develop the overall uncertainty. For simple computations a minimum of 1000 iterations can be used, however, a larger number of iterations may be necessary to determine whether the tendency of the distribution is normal or lognormal.

5.1.2.2 Method

Since different exposure geometries are more appropriate for different dose components, the individual dose components (dosimeter dose, missed dose, occupational medical dose, and environmental dose) each must be converted to organ dose. The total radiation energy interval uncertainty is then calculated by sampling from each of the organ dose distributions.

5.1.2.3 Example

Using the data from example 5.1.1.3, the total uncertainty can be computed as shown in Figure 5.1.

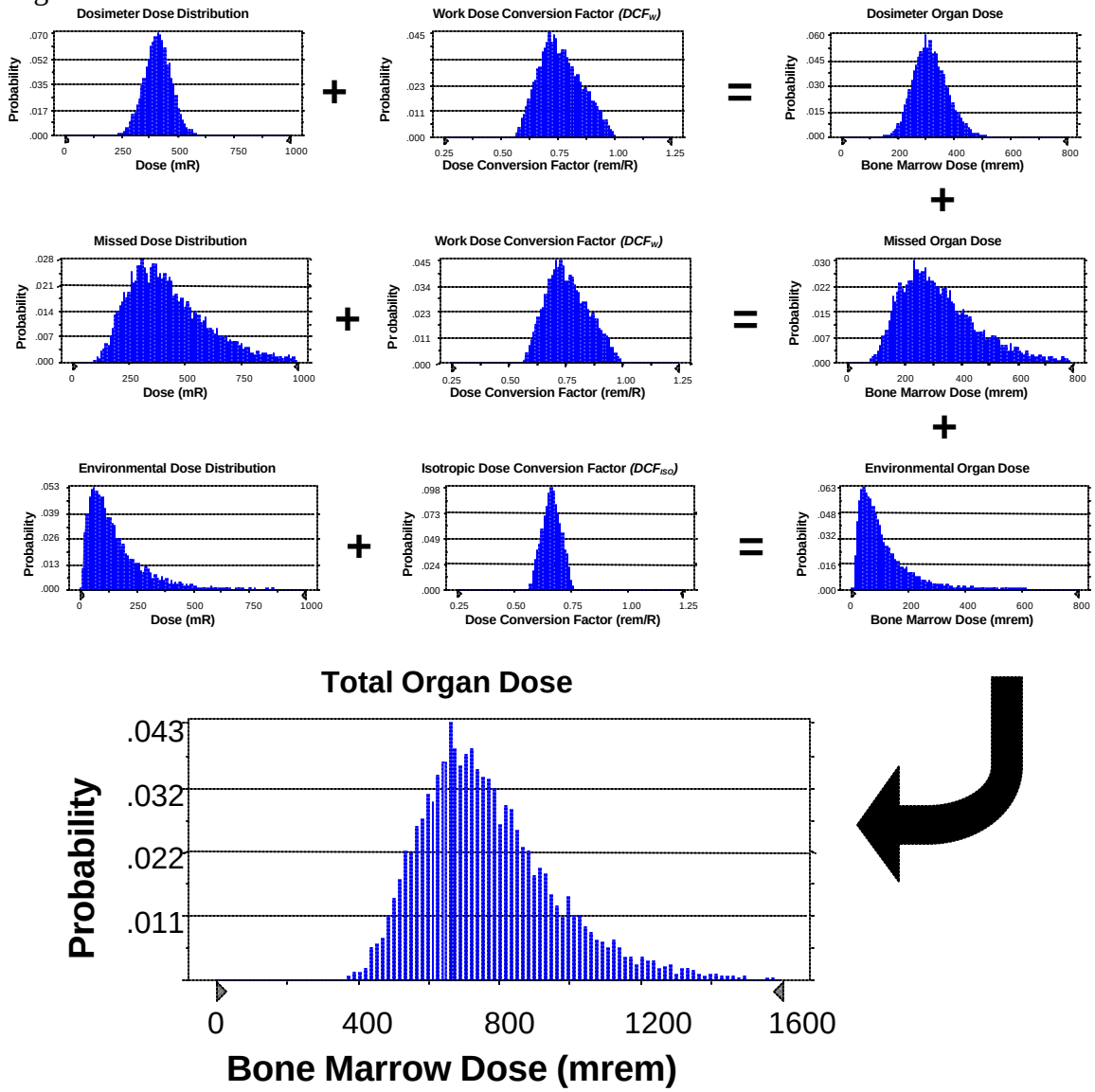

Figure 5.1 Uncertainty distribution for red bone marrow example combining organ dose uncertainties from the dosimeter dose, missed dose, and environmental dose.

The mean of the compiled distribution is 779 ± 207 mrem. However the distribution appears to be more lognormal than normal. If the distribution is lognormal, the mean and standard deviation are inappropriate parameters to describe the underlying distribution. Transforming the data results in the development of a geometric mean of 754 mrem and a geometric standard deviation of 1.28. A statistical test is needed to catagorize the tendency of the distribution.

5.2 Dose Distribution Determination/Categorization

The compiled distribution is likely to be either normally or log normally distributed. The tendency will most likely be highly dependent on the ratio between the missed dose (log normal distribution) and the dosimeter dose (normal distribution). Therefore some statistical test should be applied to determine which distribution is more appropriate. The statistical test can be conducted manually using any variety of methods or by using standard statistical software such as SAS®, StatGraphics® or SYSTAT®. Since the sampled dose distribution is likely not to fall strictly into one distribution or another, some professional judgment should be used to determine the best fit to the data. As Kumazawa (1988) found, low level doses tend to follow a log normal distribution while higher level doses near occupational exposure limits tend to follow a normal distribution. In example 5.1.2.3, the chi-square goodness of fit statistic for non-transformed data was 1551.6, and the chi-square goodness of fit statistic was 324.3 for log transformed data. Clearly the data more closely followed a lognormal distribution with a geometric mean (GM) of 754 mrem and a geometric standard deviation (GSD) of 1.28.

5.3 IREP-Excel Reporting Format

To assist in probability of causation calculations, the annual dose information should be entered into the IREP-EXCEL spreadsheet. The format for this spreadsheet can be found in Appendix C of this guide.

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